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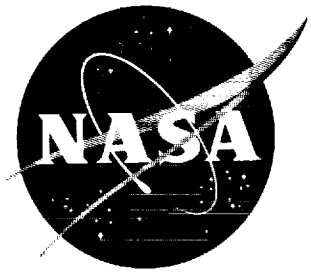

TECHNICAL NOTE

D-1322

CRITICALITY EFFECTS OF CENTRALLY LOCATED TUBES AND RODS OF ALUMINUM, IRON, AND TUNGSTEN IN A HOMOGENEOUS REACTOR

By Daniel Fieno, Eugene Gunn, Clayton Barber,
Thomas Fox, Donald Alger, and Robert Mueller

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SUMMARY

The NASA zero-power reactor, consisting of an unreflected cylinder containing a solution of uranyl fluoride salt in water, was used to study simple heterogeneous effects. This heterogeneity was introduced into the reactor as tubes or rods of aluminum, iron, and tungsten located on the axis of the reactor and extending the full height of the core. For a given hydrogen- to uranium-235-atom ratio R of the fuel solution, criticality was achieved by varying the height of the fuel solution. In addition, radial thermal-flux traverses for individual tubes of aluminum, iron, and tungsten were obtained. Some criticality measurements of cylindrical void regions located on the axis of the reactor were also made.

One-dimensional calculations with 19 energy groups were performed by using a diffusion-theory program written for the IBM 704 electronic computer. Some three-group, two-dimensional transport-theory calculations using the Los Alamos two-dimensional transport (TDC) code were also made.

Several results were established from the experiments and calculations. Cylindrical void regions and solid aluminum rods of the same diameter located at the center of the reactor with R of 300 gave nearly identical critical solution heights. The criticality of a number of inserts was adequately predicted by the Los Alamos TDC code by using three energy groups. Radial calculations using a one-dimensional diffusion-theory model can be made to agree with the experimental criticality data if the axial leakage of neutrons is properly accounted for within the insert regions. The analytical radial thermal fluxes, obtained from one-dimensional calculations for one tube each of aluminum, iron, and tungsten were in good agreement with experimental measurements. For elements with a large resonance-absorption integral, such as tungsten, the method of flux weighting proved superior to the use of the infinitely dilute resonance-absorption integral in obtaining the macroscopic absorption cross sections for the fast energy groups. The criticality of reactors with isolated, centrally located void regions was adequately described by an analytical method based on diffusion theory.

INTRODUCTION

For the past several years, conceptual studies of high-temperature nuclear reactors for space-flight applications have been made by using a hydrogenous moderator and a fuel element consisting of tungsten as the alloying and cladding material for enriched uranium dioxide fuel. Enrichment of the tungsten in the nonabsorptive W^{184} isotope is desirable. Generally, these reactors are cylindrical, with a reflector along the curved surface but not in the ends, and consist of a heterogeneous geometric arrangement of the fuel elements within the moderator. The use of hydrogenous moderators leads to reactors with predominantly thermal neutron spectra. These reactors have a number of important static-physics problems, such as (1) the streaming of neutrons of all energies through void passages, (2) the effective resonance integral of lattice arrangements of enriched tungsten and moderator (ref. 1), (3) the depression of the thermal flux in the fuel element due to the absorption of both the uranium 235 and the tungsten, and (4) the spatial hardening effects due to the absorptivity changes between the fuel and the moderator region.

The purpose of the present work was to investigate some of these effects in the NASA zero-power reactor for various inserts of tungsten, iron, and aluminum. These inserts were rods or tubes of varying sizes, extending the full height of the core and located on the axis of the reactor. The symmetry of the experimental arrangement facilitated the interpretation of the experiments. Measurements and calculations for void, hollow aluminum tubes located along the axis of the cylindrical reactor were also made.

The calculations were performed by using, primarily, a one-dimensional, multigroup, multiregion diffusion-theory code programmed for the IBM 704 electronic computer. Some results were also obtained by use of the Los Alamos two-dimensional transport code (TDC; ref. 2).

These calculations and experiments serve to evaluate various methods for obtaining macroscopic group constants for making valid parametric studies on proposed heterogeneous tungsten-fuel-element - hydrogen-moderated reactor systems.

DESCRIPTION OF CRITICALITY MEASUREMENTS WITH NASA ZERO POWER REACTOR

The NASA zero-power reactor is a thermal reactor with a homogeneous water solution of fully enriched uranyl fluoride (UO_2F_2) as fuel and moderator. The reactor, unreflected and cylindrical in shape, is

contained in a 1/8-inch-thick aluminum tank, 12 inches in inside diameter. For a given value of the hydrogen- to uranium-235-atom ratio R, criticality is achieved at a particular height of the fuel solution. Most of the experimental results were measured at a value of R of 300; the critical height of 30.73 centimeters with no inserts other than the support brackets resulted in an approximately square reactor with a length-to-diameter ratio of 1. The support brackets affected the overall solution height only slightly. At a value of R of 300, for example, the support brackets accounted for an increase in solution height from 30.56 to 30.73 centimeters. Some additional results were obtained at a value of R of 440, with a corresponding critical height of 46.68 centimeters. This fuel solution resulted in a reactor length-to-diameter ratio of approximately 1.5.

Insertion into the center of the reactor of tubes or rods that extend the full height of the core introduces the heterogeneous effect. A sketch of an absorptive tube inserted and supported in the reactor tank is shown in figure 1. With this arrangement, precise determinations may be made of critical fuel heights and radial thermal-flux distributions. The height of the reactor solution is measured by a micrometer lead screw connected to an electrical probe (see fig. 1). Changes in fuel-solution density due to evaporation and temperature, however, introduce corrections to the measurements of the solution height. After temperature and evaporation corrections have been applied, the overall solution height may be measured with an accuracy of ±0.04 centimeter. The experimental results obtained for the centrally located rods and tubes of aluminum, iron, and tungsten are summarized in table I. Some results obtained for centrally located void, hollow aluminum tubes are also given in the table. These results are given in terms of the difference between the critical heights of the reactor with insert and the reactor without insert.

Most of the inserts were coated with a thin protective layer of a plastic material, which made them resistant to chemical attack by the corrosive fuel solution. This plastic coating was about 0.005 centimeter thick, and was a significant absorber in the case of the aluminum inserts. The effect of the coating on the iron inserts was less significant since the number of absorptions in the coating was relatively small compared with the number of absorptions in the iron. The majority of the aluminum inserts were coated (see table I), but two cases obtained with uncoated tubes permitted a measurement of the worth of the coating. All the tungsten inserts were uncoated, while the iron inserts had to be protectively coated (see table I).

The radial thermal neutron flux was measured at the horizontal midplane of the reactor for an aluminum, an iron, and a tungsten tube. The thermal flux was measured through the activation of small dysprosium foils, which were 0.3175 centimeter in diameter and 0.0127 centimeter thick and were formed from a 5-percent-dysprosium - 95-percent-aluminum alloy. The foils were spaced in such a manner that they did not perturb one another.

METHODS OF CALCULATION

The theoretical calculations for the various critical experiments were based primarily on radial solutions obtained with a one-dimensional, multigroup, multiregion diffusion-theory program written for the IBM 704 electronic computer. The number of neutron energy groups used was 19. Two methods were used to estimate the transverse leakage of neutrons through the ends of the reactor. In one method, the axial leakage of neutrons from any group and region was represented by the diffusion coefficient of that group and region times the axial buckling. In the other method, the axial leakage of neutrons through the tubes or rods was considered, for any group, to be represented by the diffusion coefficient of the surrounding fuel solution times the axial buckling. In addition, some results were obtained by using the Los Alamos TDC code (ref. 2). This code is based on the S_n method of reference 2 and is written for r,z geometry, where r is the radius and z is the axial dimension. The two-dimensional calculations reported herein used three energy groups and the S_4 approximation. The parameters used for any region in the diffusion-theory code for a given neutron energy group are the diffusion coefficient D, the macroscopic absorption cross section Σ_a, the group-removal cross section Σ_q, and the production cross section $\nu\Sigma_f$. The calculation of these group parameters by use of the lethargy-dependent flux obtained for a given mixture of elements is fully described in reference 3. In this method, a sufficiently large region of the homogeneous mixture is employed to determine the lethargy-dependent flux over which to average the macroscopic cross section. The cross sections obtained in this manner for the group structures used are listed in table II for tungsten, iron, aluminum, and two solutions of water and uranyl fluoride salt (UO_2F_2). One of the solutions had an atom ratio R of 300, and the other, a value of R of 440.

In addition, calculations were performed for tungsten tubes by using the infinitely dilute resonance-absorption integral to define a macroscopic absorption cross section for tungsten; that is, the quantity Σ_a was defined as

$$\Sigma_a = \frac{N \int_{group} \sigma_a(u)\,du}{\int_{group} du} \qquad (1)$$

which corresponds to an assumption that the lethargy-dependent weighting flux used in the procedure described in reference 3 is a constant. In equation (1), the quantity $\sigma_a(u)$ is the microscopic absorption cross

section at lethargy u, where the integral of the quantity for the group in question corresponds to the contribution of the group to the infinitely dilute resonance-absorption integral. In this case, the quantity N is the atom density of tungsten in atoms per cubic centimeter. The diffusion coefficient and removal cross section for tungsten were still the flux-weighted ones previously described.

The cross sections needed for the two-dimensional r,z geometry calculations made with the TDC transport code were derived from the flux-weighted parameters. For a given group and region, the cross sections needed are the macroscopic transport cross section Σ_{tr}, the weak scattering cross section $\Sigma_{g\to g}$, the removal cross section Σ_q, and the macroscopic production cross section $\nu\Sigma_f$. The macroscopic transport cross section is taken as

$$\Sigma_{tr} = \frac{1}{3D} \tag{2}$$

A self-consistent set of group parameters is obtained if the weak scattering cross section is defined by

$$\Sigma_{g\to g} = \Sigma_{tr} - \Sigma_q - \Sigma_a \tag{3}$$

The analysis of the centrally located void regions was made by use of the method described in reference 4. In this method, the boundary condition at the interface between the void region and the fuel solution can be described in terms of effective cross sections, which depend on the length and the diameter of the void as well as on the transport mean free path of the surrounding medium. The computation of these cross sections for a given neutron energy group is given in reference 4 in terms of an effective diffusion coefficient and an absorption cross section such that the solution of a diffusion type of equation in the void matches the proper interface boundary condition.

All the calculated reactor heights computed herein for the one-dimensional radial calculations were assumed to include an energy independent 3.0-centimeter extrapolation distance on each end of the cylinder, as in reference 3. In the radial direction, the flux in each group was assumed to vanish at an extrapolation distance of 2.13 D. The diffusion coefficient D used to obtained this radial extrapolation distance was that of the aluminum tank for the given group.

RESULTS AND DISCUSSION

For a reactor with no insert, the calculated critical height based on the 19-group, one-dimensional radial-diffusion calculations was 31.72 centimeters, whereas the experimental height was 30.73 centimeters for

a reactor solution with a value of R of 300. The calculated height for the reactor with a value of R of 440 was 44.48 centimeters compared with the experimental result of 46.68 centimeters. The calculated and experimental results are given in terms of the difference in critical height ΔH of the reactor with and without an insert.

Aluminum Tubes and Rods

Calculations and experiments were performed for 4-inch-outside-diameter aluminum tubes varying in thickness from about 0.05 to 0.30 inch. Figure 2(a) gives the results for the one-dimensional calculations for the reactor containing a fuel solution with a value of R of 300. The calculations, based on the use of the diffusion coefficients of aluminum for estimating the axial leakage of neutrons for the tubes, predict a larger change in the solution height than the experiments do. Using the diffusion coefficients of the fuel solution for estimating the axial leakage through the aluminum tubes gives the dashed curve in the figure. The difference in the critical heights of the two curves represents the effects of the two methods of approximating the axial leakage. This difference represents the uncertainty of treating a basically two-dimensional problem by using separable one-dimensional axial and radial solutions. Two-dimensional transport calculations using r,z geometry properly account for axial, as well as radial, leakage. Three-group results for aluminum-tube thicknesses of 0.05 and 0.33 inch are shown in figure 2(a) and agree well with experiment.

Similar calculations and experiments for 4-inch-outside-diameter aluminum tubes are shown in figure 2(b) for a reactor fuel solution corresponding to a value of the parameter R of 440. Axial leakage of neutrons is smaller for this reactor than for the reactor with an R of 300. The effect of the protective coating is shown by the experimental points plotted in figure 2(b) and is not negligible. One-dimensional radial-diffusion calculations, based on using the diffusion coefficients of aluminum for estimating the axial leakage of neutrons for the tubes, are higher than the experimental points for uncoated aluminum tubes. Using the diffusion coefficients of the fuel solution for estimating the axial leakage through the aluminum tubes gives the results shown as the dashed curve in the figure. Three-group, two-dimensional transport-theory calculations were made for two aluminum tubes. These points are shown on figure 2(b) and the agreement with the uncoated tubes is satisfactory.

As an indication of the different sensitivity for the same insert and as a test of the analysis, the following is noted: For the reactor solution with R of 300, the 0.228-inch coated aluminum tube gave an increase in reactor height for the experiment of about 4.9 centimeters, while the corresponding change in the fuel solution with R of 440 was 16.7 centimeters.

The experimental thermal fluxes obtained for the 0.117-inch aluminum annulus with thin dysprosium foils and for the reactor solution at R of 300 are plotted in figure 3. The solid curve is the radial flux obtained from a 19-group calculation using flux-weighted aluminum cross sections. The experiments were normalized to the calculated thermal flux obtained at the center of the reactor. The agreement of the calculated thermal flux with experiment is good.

The experimental results obtained for aluminum rods inserted on the axis of the reactor containing a solution with an atom ratio R of 300 are given in figure 4. The aluminum rods varied from about 5/8 to $2\frac{1}{2}$ inches in diameter and were all protected by the coating. The solid analytical curve is higher than the experimental points as it was for the aluminum tubes. This curve is based on one-dimensional radial calculations and on an estimation of the axial leakage of neutrons through the rods by using the diffusion coefficients of aluminum. The use of the diffusion coefficients of the fuel solution for estimating the axial leakage through the aluminum rods gives the dashed curve shown in the figure. A three-group, two-dimensional transport calculation for the 2-inch-diameter rod is also shown in figure 4. All calculations are for uncoated rods.

The curves shown in figures 2 and 4 show that the calculated changes in the reactor height for the aluminum inserts considered are sensitively dependent on the estimate of the axial neutron leakage. Estimates of the axial leakage of neutrons through the aluminum regions based on the diffusion coefficients of aluminum are higher than the experimental points for the uncoated inserts. Calculations based on an estimation of the axial leakage of neutrons through the aluminum regions by using the diffusion coefficients of the fuel solution have been more nearly in agreement with the experiments. The few two-dimensional calculations based on transport theory that correctly account for the loss of neutrons through the boundaries of the reactor gave satisfactory agreement with the uncoated experimental results.

Void Regions in Hollow Aluminum Tubes

Calculations and experiments were also made for thin, void, hollow aluminum tubes inserted on the axis of the research reactor using the fuel solution with a value of the parameter R of 300 (fig. 5). The tube walls were about 0.03 inch thick and the tube outside surfaces, which were in contact with the fuel solution, were protectively coated. The hollow aluminum tubes varied from about 1/2 to $1\frac{1}{2}$ inches in diameter. Also shown in the figure are the experimental results for the aluminum rods. Over the range covered, the aluminum rods and void aluminum tubes

gave nearly identical experimental results. A curve based on one-dimensional, radial, 19-energy-group calculations based on the method described in reference 4 is also shown. In this diffusion-theory method, the streaming of neutrons through an isolated void located on the axis of a cylinder can be described in terms of a boundary condition at the interface of the void for each of the neutron groups. This boundary condition, in turn, can be used to define for the void region and a given neutron energy group an effective diffusion coefficient and macroscopic absorption cross section based on the length and the diameter of the void and on the transport mean free path of the surrounding medium. The agreement of this calculation with experiment is satisfactory. Another calculation was made with the assumption that the neutrons streaming through the cylindrical void could be described solely in terms of a leakage loss based on the diffusion coefficients of the surrounding fuel solution and on the axial buckling of the reactor. This calculation is shown as the dashed curve in figure 5.

Iron Tubes and Rods

Calculations and experiments were performed for 4-inch-outside-diameter iron tubes varying in thickness from about 0.069 to 0.228 inch. All the iron tubes and rods were protectively coated. Because of the increased absorptions of the iron as compared with aluminum, this coating caused only a small fraction of the change in fuel-solution height in the reactor with and without an iron insert. The results for the one-dimensional calculations for the reactor containing a fuel solution with a value of R of 300 are given in figure 6(a). The calculations, based on using the diffusion coefficients of iron for estimating the axial leakage of neutrons for the tubes, predict a somewhat larger change in the solution height than occurred in the experiments. Using the diffusion coefficients of the fuel solution for estimating the axial leakage through the iron tubes gives the dashed curve in the figure. The difference between the two methods of calculating the axial leakage through the iron tubes is slight. A three-group, two-dimensional transport calculation using r,z geometry was made for the 0.228-inch iron tube and agreed with the experimental value for the change in fuel-solution height.

Similar calculations and one experimental point for 4-inch-outside-diameter iron tubes are shown in figure 6(b) for a reactor fuel solution corresponding to a value of the parameter R of 440. The calculations based on the two methods of treating the axial leakage of neutrons gave nearly identical results. The experimental point is about 4 centimeters higher than the corresponding analytical point.

In figure 7 are plotted experimental values of the radial thermal flux obtained for a 0.121-inch-thick, 4-inch-outside-diameter iron tube in a reactor fuel solution with an R of 300. The solid curve is based

on 19-group diffusion-theory calculations using flux-weighted cross sections. The experiments were normalized to the calculated thermal flux obtained at the center of the reactor. The agreement of the calculated thermal flux with experiment is good.

Experimental results were obtained for coated iron rods inserted on the axis of the reactor containing a fuel solution having a value of the parameter R of 300 (fig. 8). The iron rods varied from 1/4 to 2 inches in diameter. The solid analytical curve is somewhat higher than the experimental points as it was for the iron tubes. This curve is based on one-dimensional radial calculations and on the estimation of the axial leakage of neutrons through the rods by using the diffusion coefficients of iron. Using the diffusion coefficients of the fuel solution for estimating the axial leakage through the iron rods gives the dashed curve shown in the figure. The difference between the two methods of calculating the axial leakage of neutrons through the iron rods is slight.

Tungsten Tubes

Calculations and experiments were made for 4-inch-outside-diameter natural tungsten tubes varying in thickness from 0.01 to 0.05 inch. The reactor vessel contained fuel solution with a value of R of 300. Figure 9 gives the results for one-dimensional radial-diffusion calculations based on 19 energy groups. The solid curve in the figure corresponds to the use of flux-weighted tungsten cross sections. The agreement between experiment and calculation is good. The dashed curve in the figure corresponds to 19-group calculations based on using the dilute resonance-integral data to obtain the macroscopic tungsten absorption cross sections. This curve gives poor agreement with both the flux-weighted calculations and the experiments because flux depression and self-shielding in resonances are ignored.

The experimental values of the radial thermal flux obtained for a 0.02-inch-thick, 4-inch-outside-diameter tungsten tube in a reactor solution with an R of 300 are shown in figure 10. The solid curve is based on a 19-group diffusion-theory calculation using flux-weighted cross sections. The experiments were normalized to the calculated value of the thermal flux obtained at the center of the reactor. The thermal flux near the absorbing annulus shows considerable flux depression, but the agreement of the calculated to the measured flux is good. Thus, for tungsten, which has a large thermal absorption cross section (19.2 barns) and absorption-resonance integral (350 barns), the calculated thermal neutron flux distribution is in good agreement with the experimental measurements.

CONCLUDING REMARKS

The NASA zero-power, homogeneous, solution reactor was used to obtain criticality data for inserts of aluminum, iron, and tungsten. For aluminum and iron, these inserts were in the form of tubes and rods located on the axis of the reactor and extending the full height of the core. The tungsten was used in the form of tubes. Some criticality measurements for void, hollow aluminum tubes were also made. The calculations and experiments evaluated various methods for obtaining macroscopic group constants for making valid parametric studies on proposed heterogeneous tungsten-fuel-element - hydrogen-moderated reactor systems. Sufficient results were obtained to make the following conclusions:

1. Cylindrical void regions and solid aluminum rods of the same diameter located at the center of the reactor with a hydrogen- to uranium-235-atom ratio of 300 gave nearly identical critical solution heights.

2. The criticality of a number of inserts was adequately predicted by the Los Alamos two-dimensional transport code using three energy groups.

3. Radial calculations using a one-dimensional diffusion-theory model can be made to agree with the experimental criticality data if the axial leakage of neutrons is properly accounted for within the insert region. This problem can be important for aluminum regions.

4. The axial leakage of neutrons through the insert regions is better described, especially for aluminum, by ascribing the diffusion properties of the surrounding fuel solution to the insert rather than by using the properties of the insert.

5. The analytical radial thermal fluxes, obtained from one-dimensional calculations for one tube each of aluminum, iron, and tungsten were in good agreement with experimental measurements.

6. For elements with a large resonance-absorption integral, such as tungsten, care must be used in obtaining macroscopic absorption cross sections for each fast neutron energy group. The method of flux weighting proved superior to the use of the infinitely dilute resonance-absorption integral. Calculation of the critical solution height for the tungsten tubes by using flux-weighted cross sections was in good agreement with the experimental results. The calculated radial thermal-flux distribution was in good agreement with experiment for the 0.02-inch-wall-thickness 4-inch-outside-diameter tungsten tube.

7. An analytical method based on diffusion theory was adequate for predicting the criticality of reactors with isolated, centrally located void regions.

Lewis Research Center
National Aeronautics and Space Administration
Cleveland, Ohio, June 7, 1962

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TABLE I. - NASA ZERO-POWER-REACTOR EXPERIMENTAL CRITICALITY MEASUREMENTS

Material description	Critical height with insert minus critical height without insert, ΔH, cm
Hydrogen- to uranium-235-atom ratio, 300; critical height without insert, 30.73 cm	
Aluminum	
4-in.-O.D. tubes	
0.052-in. wall, uncoated	0.94
0.052-in. wall, coated	1.17
0.117-in. wall, coated	2.58
0.228-in. wall, coated	4.91
0.332-in. wall, coated	7.25
Rods	
5/8-in. diam., coated	.57
1-in. diam., coated	1.49
$1\frac{1}{4}$- in. diam., coated	2.36
$1\frac{1}{2}$- in. diam., coated	3.58
2-in. diam., coated	6.49
$2\frac{1}{2}$- in. diam., coated	10.92
Hollow tubes, 0.030-in. wall	
1/2-in. O.D., coated	.39
1-in. O.D., coated	1.59
$1\frac{1}{2}$- in. O.D., coated	3.63
Iron	
4-in.-O.D. tubes	
0.069-in. wall, coated	3.71
0.121-in. wall, coated	6.49
0.186-in. wall, coated	10.15
0.228-in. wall, coated	13.52
Rods	
1/4-in. diam., coated	.22
1/2-in. diam., coated	.79
3/4-in. diam., coated	1.67
1-in. diam., coated	2.77
$1\frac{1}{2}$- in. diam., coated	5.80
2-in. diam., coated	10.50
Tungsten	
4-in.-O.D. tubes	
0.010-in. wall, uncoated	3.10
0.020-in. wall, uncoated	5.70
0.050-in. wall, uncoated	[a]13.5
Hydrogen- to uranium-235-atom ratio, 440; critical height without insert, 46.68 cm	
Aluminum	
4-in.-O.D. tubes	
0.052-in. wall, uncoated	2.67
0.228-in. wall, uncoated	15.19
0.052-in. wall, coated	3.65
0.117-in. wall, coated	8.20
0.228-in. wall, coated	16.68
Iron	
4-in.-O.D. tube	
0.069-in. wall, coated	18.50

[a]Estimated from neutron multiplication data.

TABLE II. - MACROSCOPIC NEUTRON CROSS SECTIONS FOR ALUMINUM, IRON, TUNGSTEN, AND REACTOR FUEL SOLUTION (URANYL FLUORIDE AND WATER) WITH HYDROGEN- TO URANIUM-235-ATOM RATIOS OF 300 AND 440

(a) Three-group flux-weighted cross sections

Lethargy range	Diffusion coefficient D, cm	Macroscopic absorption cross section, Σ_a, cm^{-1}	Group-removal cross section, Σ_q, cm^{-1}	Production cross section, $\nu\Sigma_f$, cm^{-1}
Reactor fuel solution (uranyl fluoride and water); hydrogen- to uranium-235-atom ratio, 300				
0.00- 7.75	1.509019	0.000528	0.065313	0.000942
7.75-18.20	.592768	.014918	.118104	.023651
19.80	.143681	.134740	.000000	.244039
Reactor fuel solution (uranyl fluoride and water); hydrogen- to uranium-235-atom ratio, 440				
0.00- 7.75	1.516005	0.000396	0.066364	0.000646
7.75-18.20	.592785	.010873	.121887	.016461
19.80	.146871	.101819	.000000	.174392
Aluminum				
0.00- 7.75	4.925644	0.000136	0.001307	--------
7.75-18.20	4.013049	.000656	.000104	--------
19.80	4.041678	.012270	.000000	--------
Iron				
0.00- 7.75	2.032850	0.000178	0.002136	--------
7.75-18.20	.447020	.004781	.000019	--------
19.80	.291170	.190129	.000000	--------
Tungsten				
0.00- 7.75	0.586563	0.012893	0.000306	--------
7.75-18.20	.412299	.127609	.000026	--------
19.80	.217945	.995483	.000000	--------

(b) Three-group resonance-integral cross sections

Tungsten				
0.00- 7.75	0.586560	0.013532	0.000306	--------
7.75-18.20	.412300	2.095500	.000026	--------
19.80	.217940	.995483	.000000	--------

TABLE II. - Continued. MACROSCOPIC NEUTRON CROSS SECTIONS FOR ALUMINUM, IRON, TUNGSTEN, AND REACTOR FUEL SOLUTION (URANYL FLUORIDE AND WATER) WITH HYDROGEN- TO URANIUM-235-ATOM RATIOS OF 300 AND 440

(c) 19-Group flux-weighted cross sections

Lethargy range	Diffusion coefficient D, cm	Macroscopic absorption cross section, Σ_a, cm^{-1}	Group-removal cross section, Σ_q, cm^{-1}	Production cross section, $\nu\Sigma_f$, cm^{-1}
Reactor fuel solution (uranyl fluoride and water); hydrogen- to uranium-235-atom ratio, 300				
0.00- 1.00	2.764563	0.001498	0.098397	0.000920
1.00- 2.00	2.119447	.000310	.139287	.000824
2.00- 3.00	1.415712	.000296	.266687	.000700
3.00- 4.00	1.130951	.000343	.507211	.000745
4.00- 5.00	1.018243	.000449	.794176	.000939
5.00- 6.00	.864428	.000632	1.066702	.001269
6.00- 7.00	.760874	.000946	1.230598	.001770
7.00- 8.00	.691769	.001436	1.304673	.002500
8.00- 9.00	.648156	.001971	1.336598	.003335
9.00-10.00	.623041	.003856	1.351543	.006126
10.00-11.00	.608655	.006388	1.355529	.009379
11.00-12.00	.598798	.008808	1.363646	.012692
12.00-13.00	.586471	.016160	1.375146	.024734
13.00-14.00	.580163	.025794	1.384991	.036838
14.00-15.00	.578348	.013154	1.387100	.013513
15.00-16.00	.575947	.009690	1.397079	.013498
16.00-17.00	.565724	.022986	1.441248	.039491
17.00-18.20	.508647	.056327	1.478254	.104933
19.80	.143681	.134740	0.000000	.244039
Reactor fuel solution (uranyl fluoride and water); hydrogen- to uranium-235-atom ratio, 440				
0.00- 1.00	2.777330	0.001381	0.098522	0.000630
1.00- 2.00	2.128410	.000213	.139701	.000564
2.00- 3.00	1.421880	.000203	.267602	.000479
3.00- 4.00	1.135870	.000235	.508614	.000510
4.00- 5.00	1.020540	.000307	.796026	.000643
5.00- 6.00	0.865640	.000433	1.069347	.000870
6.00- 7.00	.760860	.000652	1.234097	.001213
7.00- 8.00	.691230	.001005	1.308370	.001712
8.00- 9.00	.647460	.001372	1.340442	.002284
9.00-10.00	.622290	.002683	1.355291	.004197
10.00-11.00	.607970	.004446	1.358979	.006426
11.00-12.00	.598330	.006145	1.366765	.008695
12.00-13.00	.587800	.011262	1.377144	.016955
13.00-14.00	.581990	.018116	1.385877	.025479
14.00-15.00	.579540	.009565	1.389551	.009312
15.00-16.00	.576420	.007465	1.400972	.009262
16.00-17.00	.566920	.017098	1.443290	.027058
17.00-18.20	.511650	.040912	1.479011	.071904
19.80	.146870	.101819	0.000000	.174392

Lethargy range	Diffusion coefficient D, cm	Macroscopic absorption cross section, Σ_a, cm^{-1}	Group-removal cross section, Σ_q, cm^{-1}
Aluminum			
0.00- 1.00	3.775443	0.001476	0.051309
1.00- 2.00	3.338501	.000071	.022937
2.00- 3.00	2.509605	.000060	.014744
3.00- 4.00	2.031439	.000071	.014321
4.00- 5.00	2.383977	.000201	.016187
5.00- 6.00	7.127047	.000202	.007145
6.00- 7.00	6.440068	.000146	.003900
7.00- 8.00	3.927609	.000060	.006344
8.00- 9.00	4.050749	.000060	.006051
9.00-10.00	4.049617	.000077	.006038
10.00-11.00	4.045916	.000134	.006013
11.00-12.00	4.040339	.000219	.005971
12.00-13.00	4.031119	.000362	.005904
13.00-14.00	4.016560	.000587	.005795
14.00-15.00	3.991349	.000982	.005608
15.00-16.00	3.951688	.001612	.005320
16.00-17.00	3.888601	.002636	.004911
17.00-18.20	3.779903	.004489	.003171
19.80	4.041678	.012270	.000000
Iron			
0.00- 1.00	1.978067	0.000589	0.112912
1.00- 2.00	1.890407	.000085	.068678
2.00- 3.00	1.773797	.000085	.031081
3.00- 4.00	1.648244	.000085	.011922
4.00- 5.00	1.364358	.000100	.010857
5.00- 6.00	3.884298	.000181	.011638
6.00- 7.00	1.974838	.000256	.008704
7.00- 8.00	0.619518	.000473	.019763
8.00- 9.00	.605012	.000750	.019492
9.00-10.00	.414768	.001243	.028169
10.00-11.00	.359370	.002062	.032083
11.00-12.00	.354706	.003411	.031787
12.00-13.00	.358760	.005593	.030235
13.00-14.00	.356917	.009191	.028518
14.00-15.00	.353943	.015032	.025801
15.00-16.00	.349267	.024477	.021867
16.00-17.00	.341990	.039625	.017222
17.00-18.20	.330464	.065060	.006252
19.80	.291172	.190129	.000000
Tungsten			
0.00- 1.00	0.962509	0.000870	0.161702
1.00- 2.00	.766009	.001637	.233340
2.00- 3.00	.755147	.003083	.126289
3.00- 4.00	.669249	.005680	.030586
4.00- 5.00	.586897	.009660	.008472
5.00- 6.00	.535086	.016680	.004632
6.00- 7.00	.452102	.030622	.005540
7.00- 8.00	.389726	.047518	.008391
8.00- 9.00	.367049	.061779	.007597
9.00-10.00	.318664	.083494	.008030
10.00-11.00	.253598	.173502	.005226
11.00-12.00	.497079	.048772	.010057
12.00-13.00	.309764	.136959	.007930
13.00-14.00	.410349	.399386	.006679
14.00-15.00	.516423	.419832	.002607
15.00-16.00	.666684	.199453	.002524
16.00-17.00	.609093	.236743	.001776
17.00-18.20	.481354	.383038	.000677
19.80	.217945	.995483	.000000

TABLE II. - Concluded. MACROSCOPIC NEUTRON CROSS SECTIONS FOR ALUMINUM, IRON, TUNGSTEN, AND REACTOR FUEL SOLUTION (URANYL FLUORIDE AND WATER) WITH HYDROGEN- TO URANIUM-235-ATOM RATIOS OF 300 AND 440

(d) 19-Group resonance-integral cross sections

Lethargy range	Diffusion coefficient, D, cm	Macroscopic absorption cross section, Σ_a, cm^{-1}	Group-removal cross section, Σ_q, cm^{-1}
Tungsten			
0.00- 1.00	0.962510	0.000866	0.161702
1.00- 2.00	.766010	.001584	.233340
2.00- 3.00	.755150	.002895	.126289
3.00- 4.00	.669250	.005287	.030586
4.00- 5.00	.586900	.009625	.008472
5.00- 6.00	.535090	.017593	.004632
6.00- 7.00	.452100	.032232	.005540
7.00- 8.00	.389730	.047953	.008391
8.00- 9.00	.367050	.062094	.007597
9.00-10.00	.318660	.084088	.008030
10.00-11.00	.253600	.278210	.005226
11.00-12.00	.497080	.192130	.010057
12.00-13.00	.309760	1.657600	.007930
13.00-14.00	.410350	11.720000	.006679
14.00-15.00	.516430	6.976600	.002607
15.00-16.00	.666690	.201740	.002524
16.00-17.00	.609090	.239430	.001776
17.00-18.20	.481350	.393550	.000677
19.80	.217940	.995483	.000000

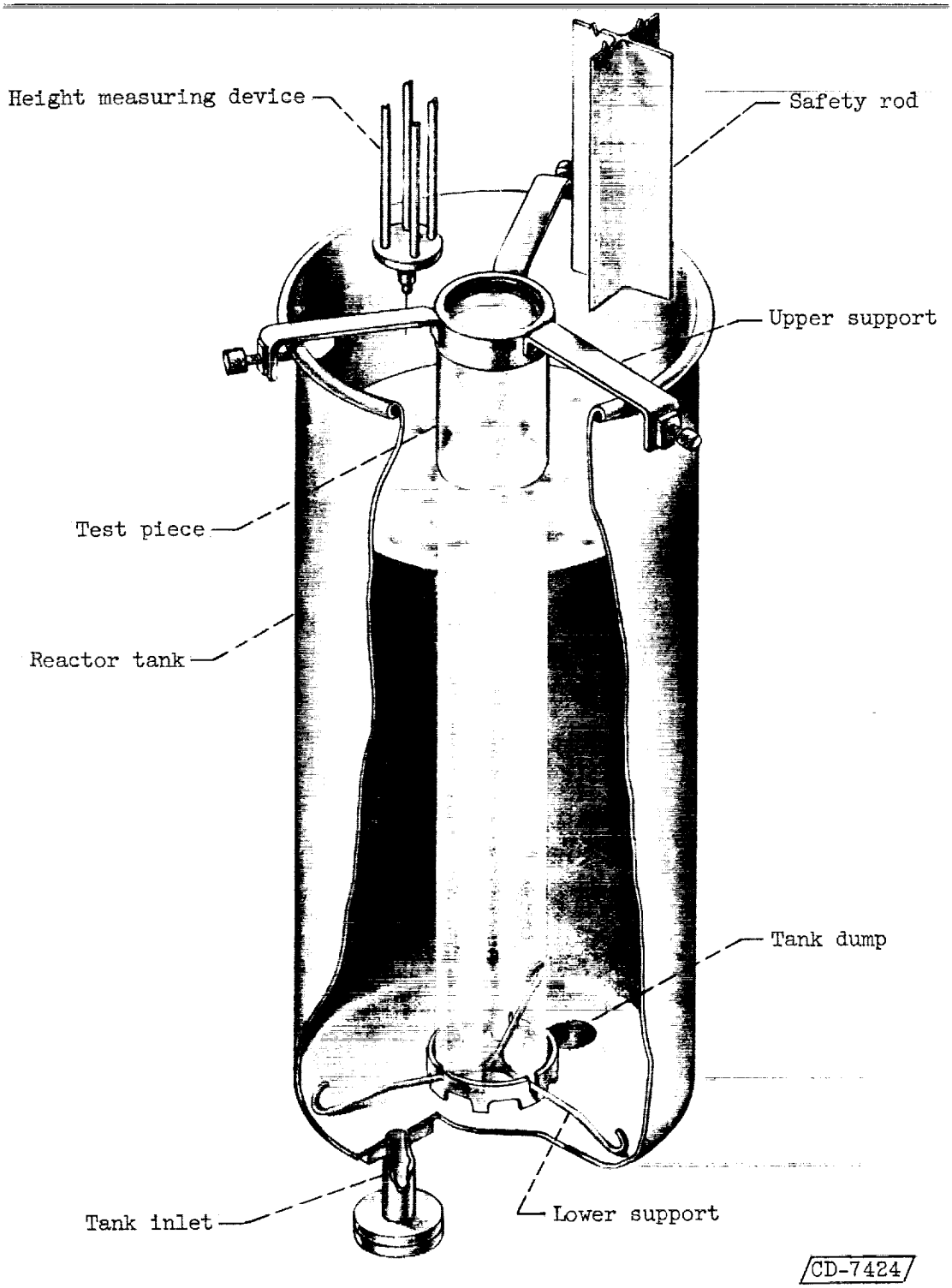
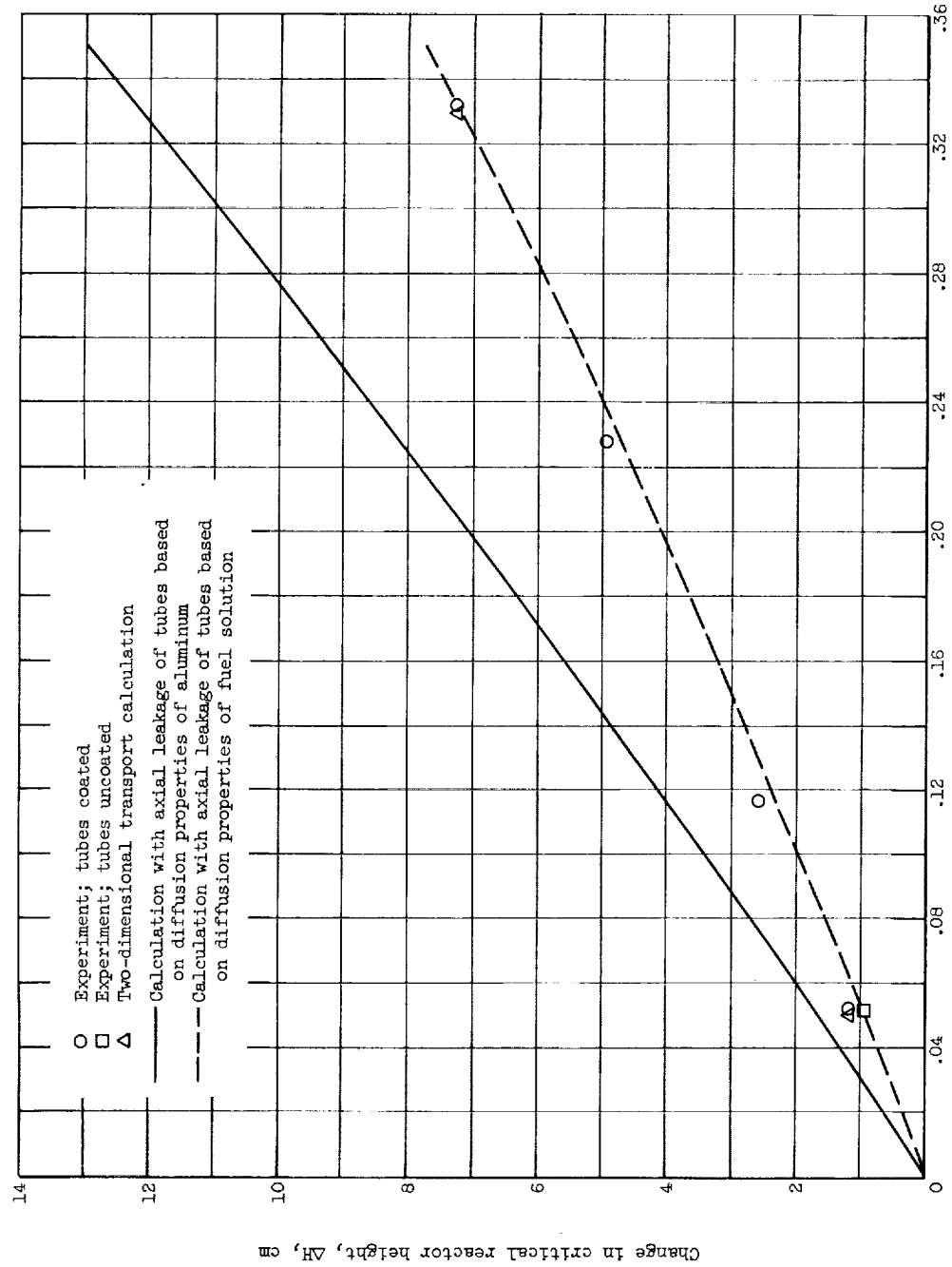
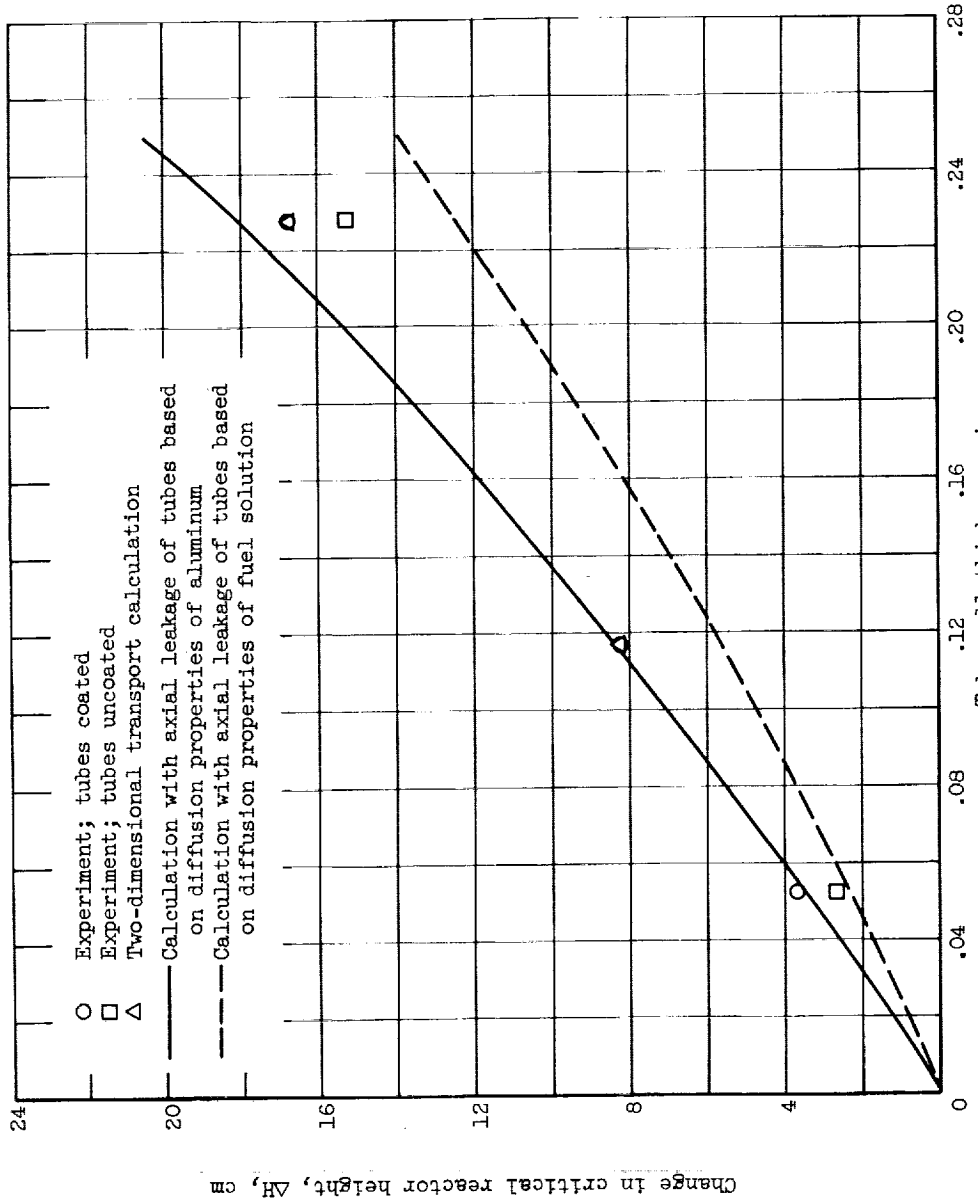


Figure 1. - Reactor containing solution of uranyl fluoride and water with annular insert region.



(a) Critical height with no insert, 30.73 centimeters; fuel solution at hydrogen- to uranium-235-atom ratio of 300.

Figure 2. - Change in critical reactor height as function of wall thickness of 4-inch-outside-diameter aluminum-tube inserts.



(b) Critical height with no insert, 46.68 centimeters; fuel solution at hydrogen- to uranium-235-atom ratio of 440.

Figure 2. - Concluded. Change in critical reactor height as function of wall thickness of 4-inch-outside-diameter aluminum-tube inserts.

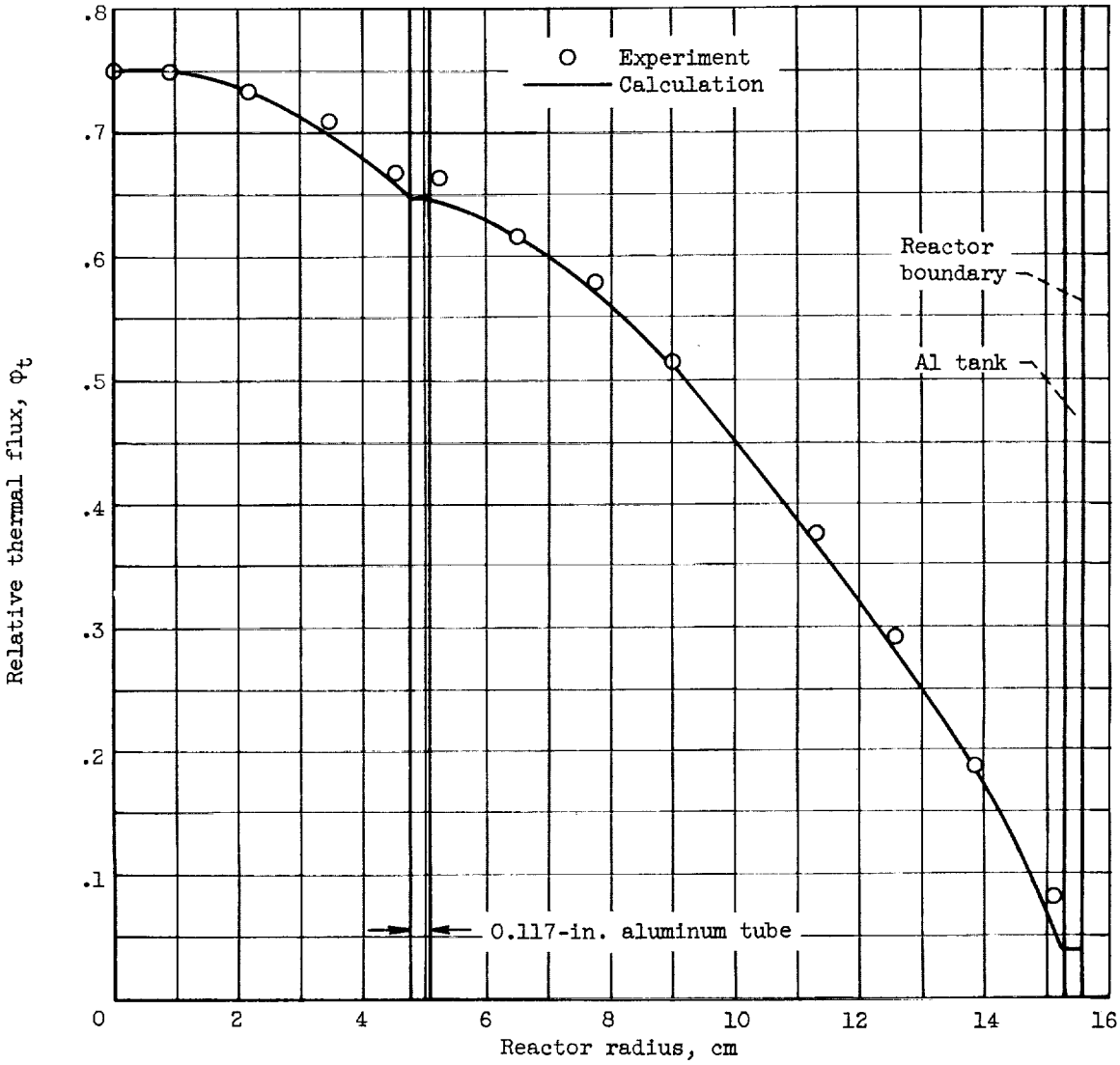


Figure 3. - Relative thermal flux as function of reactor radius with 4-inch-outside-diameter aluminum-tube insert of 0.117-inch wall thickness. Fuel solution at hydrogen- to uranium-235-atom ratio of 300.

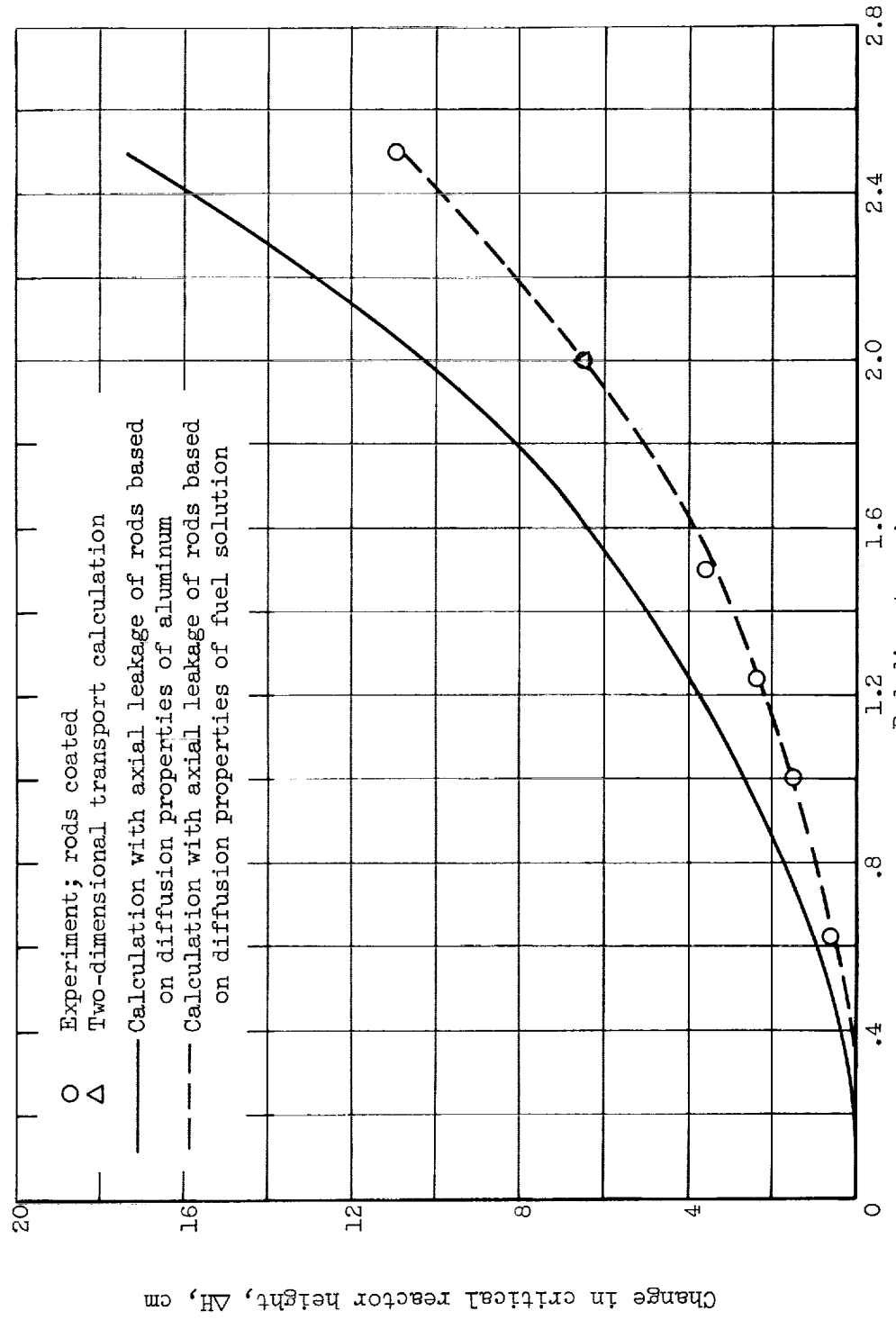


Figure 4. - Change in critical reactor height as function of diameter of aluminum-rod inserts. Critical height with no insert, 30.73 centimeters; fuel solution at hydrogen- to uranium-235-atom ratio of 300.

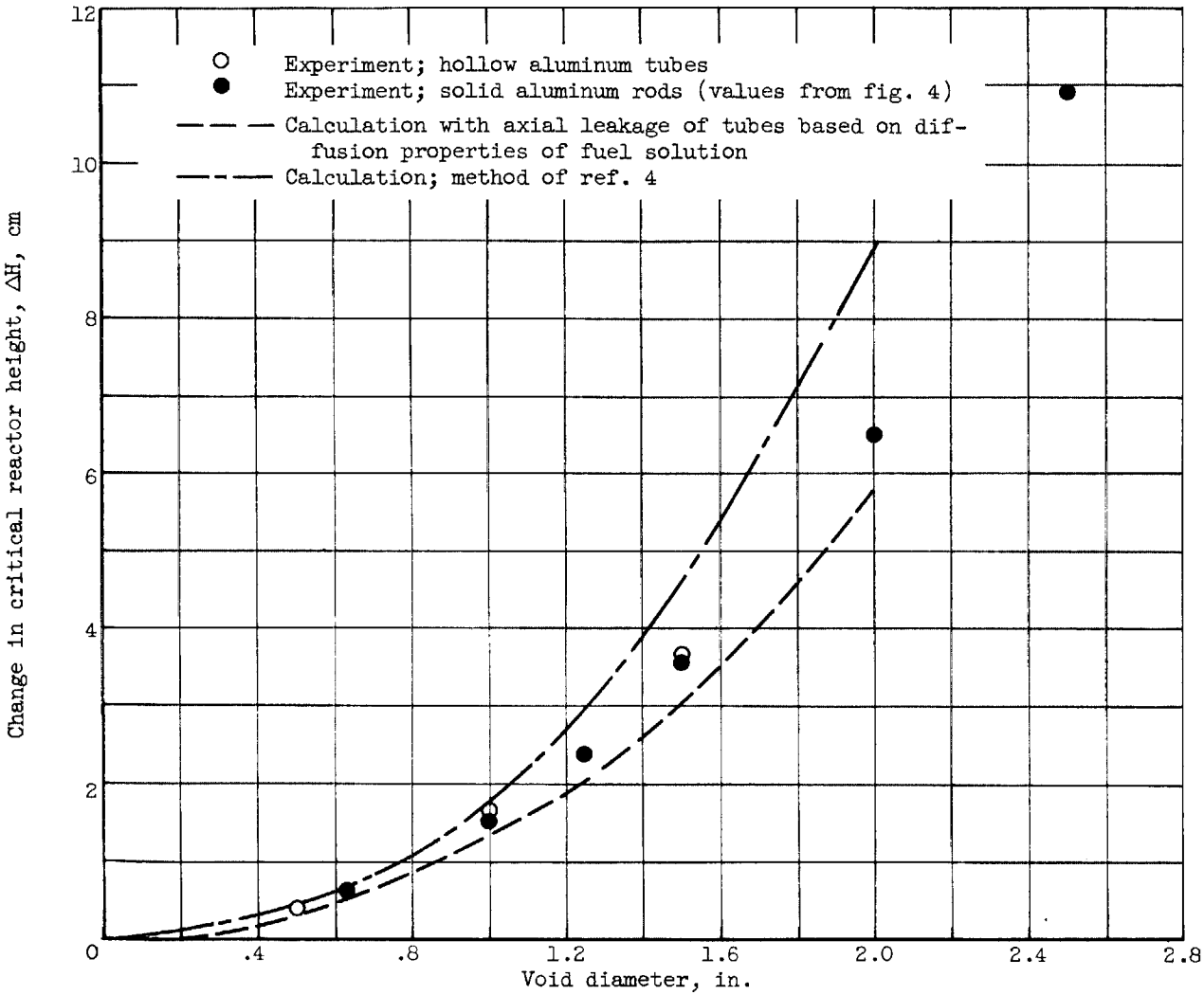
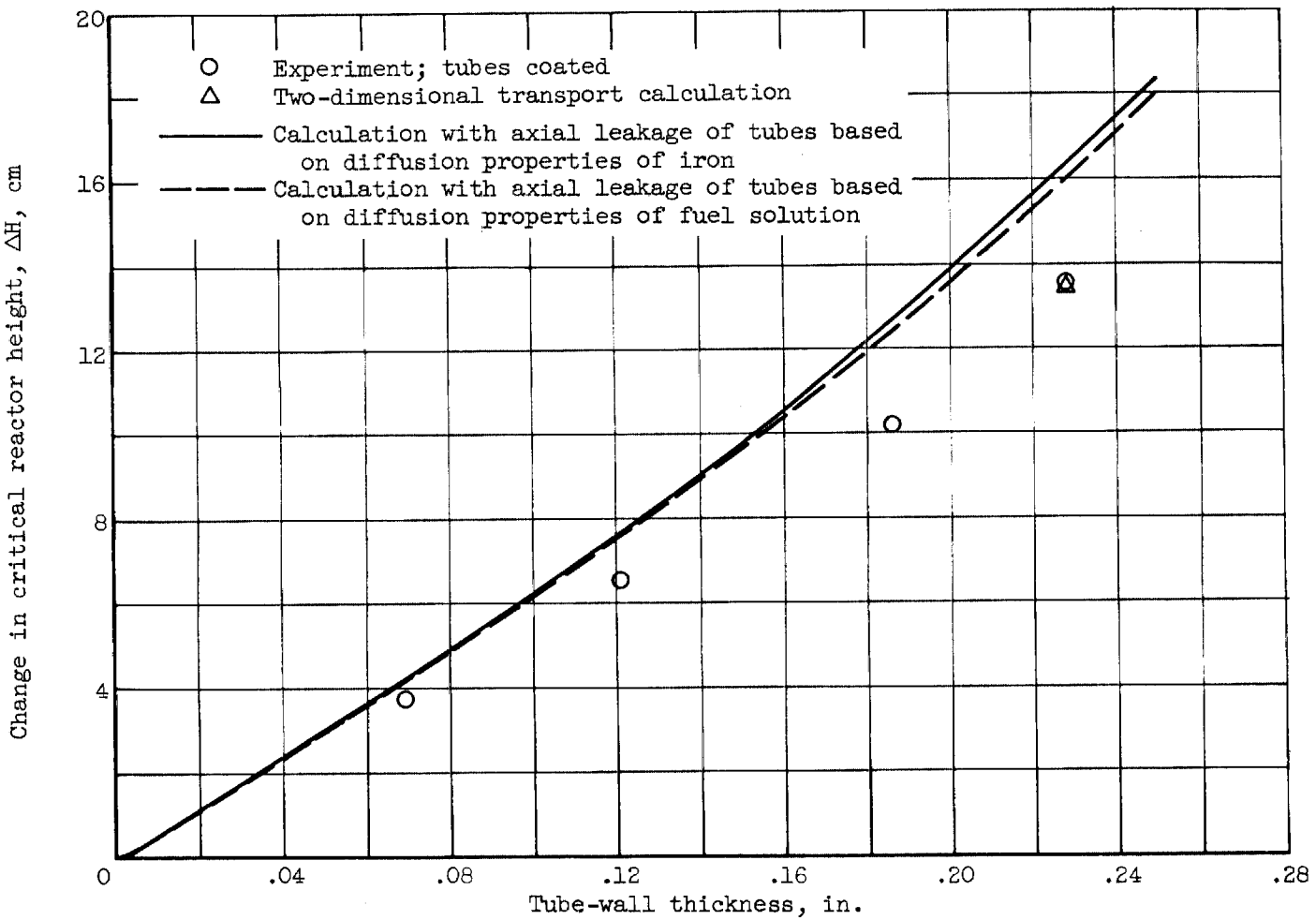
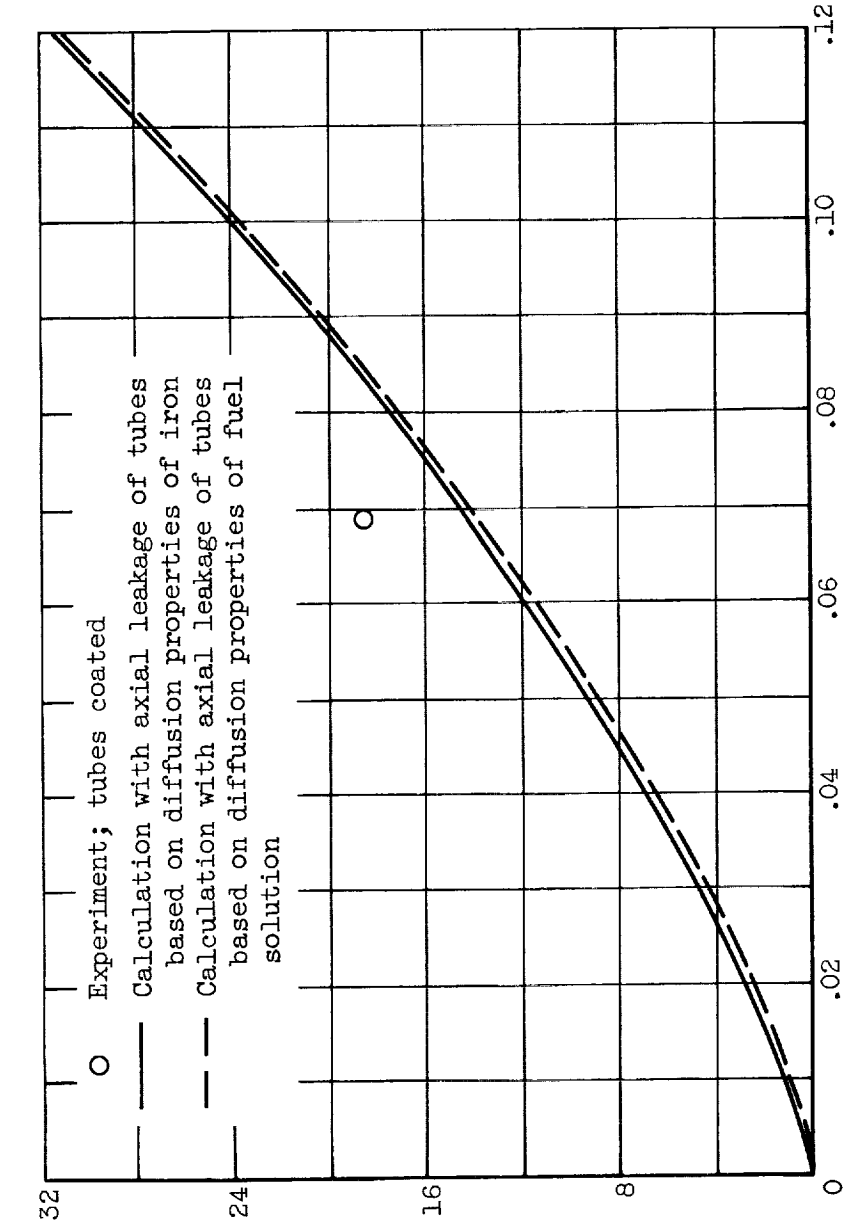


Figure 5. - Change in critical reactor height as function of diameter of inserted void region. Critical height with no insert, 30.73 centimeters; fuel solution at hydrogen- to uranium-235-atom ratio of 300.



(a) Critical height with no insert, 30.73 centimeters; fuel solution at hydrogen- to uranium-235-atom ratio of 300.

Figure 6. - Change in critical reactor height as function of wall thickness of 4-inch-outside-diameter iron-tube inserts.



(b) Critical height with no insert, 46.68 centimeters; fuel solution at hydrogen- to uranium-235-atom ratio of 440.

Figure 6. - Concluded. Change in critical reactor height as function of wall thickness of 4-inch-outside-diameter iron-tube inserts.

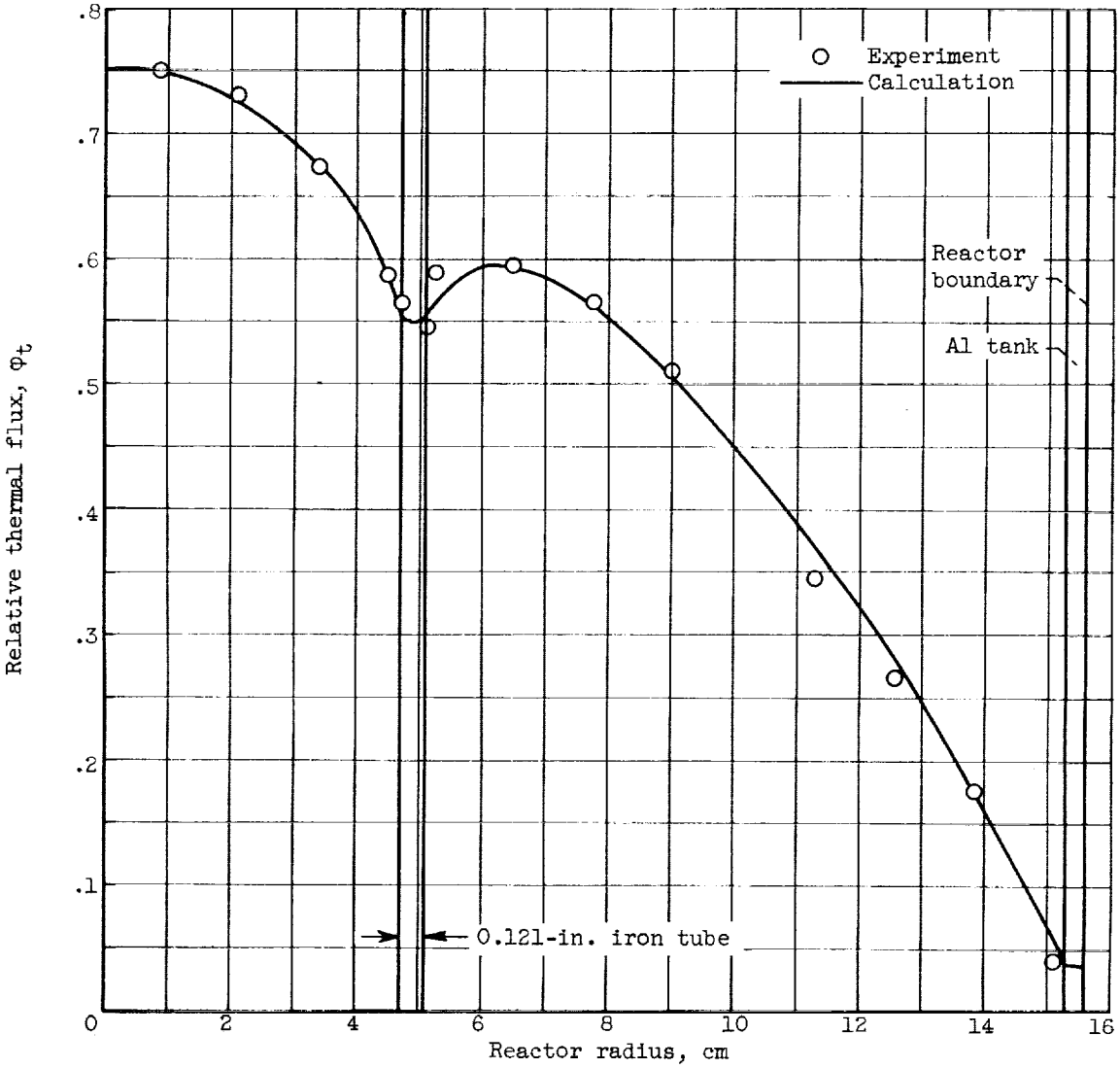


Figure 7. - Relative thermal flux as function of reactor radius with 4-inch-outside-diameter iron-tube insert of 0.121-inch wall thickness. Fuel solution at hydrogen- to uranium-235-atom ratio of 300.

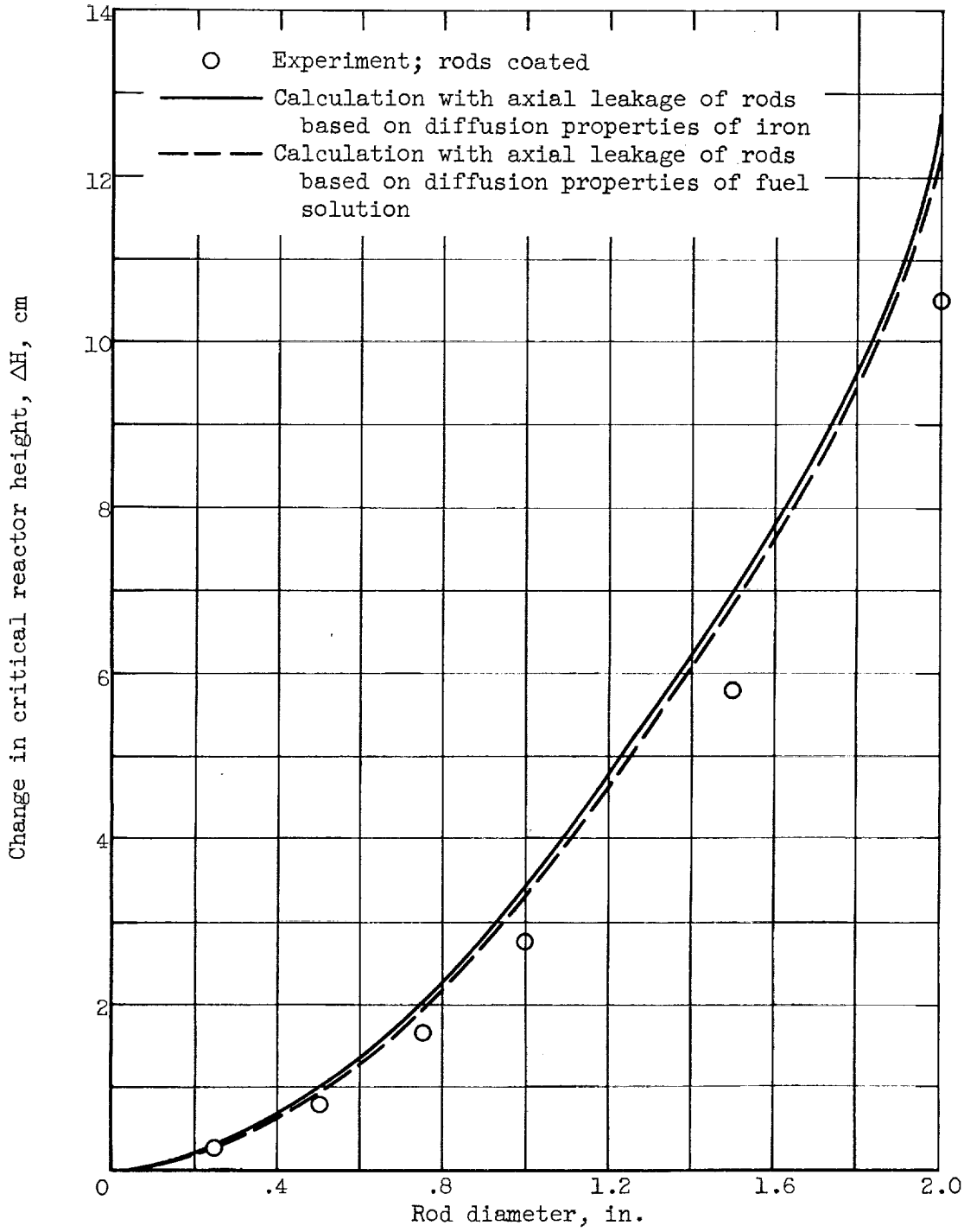


Figure 8. - Change in critical reactor height as function of diameter of iron-rod inserts. Critical height with no insert, 30.73 centimeters; fuel solution at hydrogen- to uranium-235-atom ratio of 300.

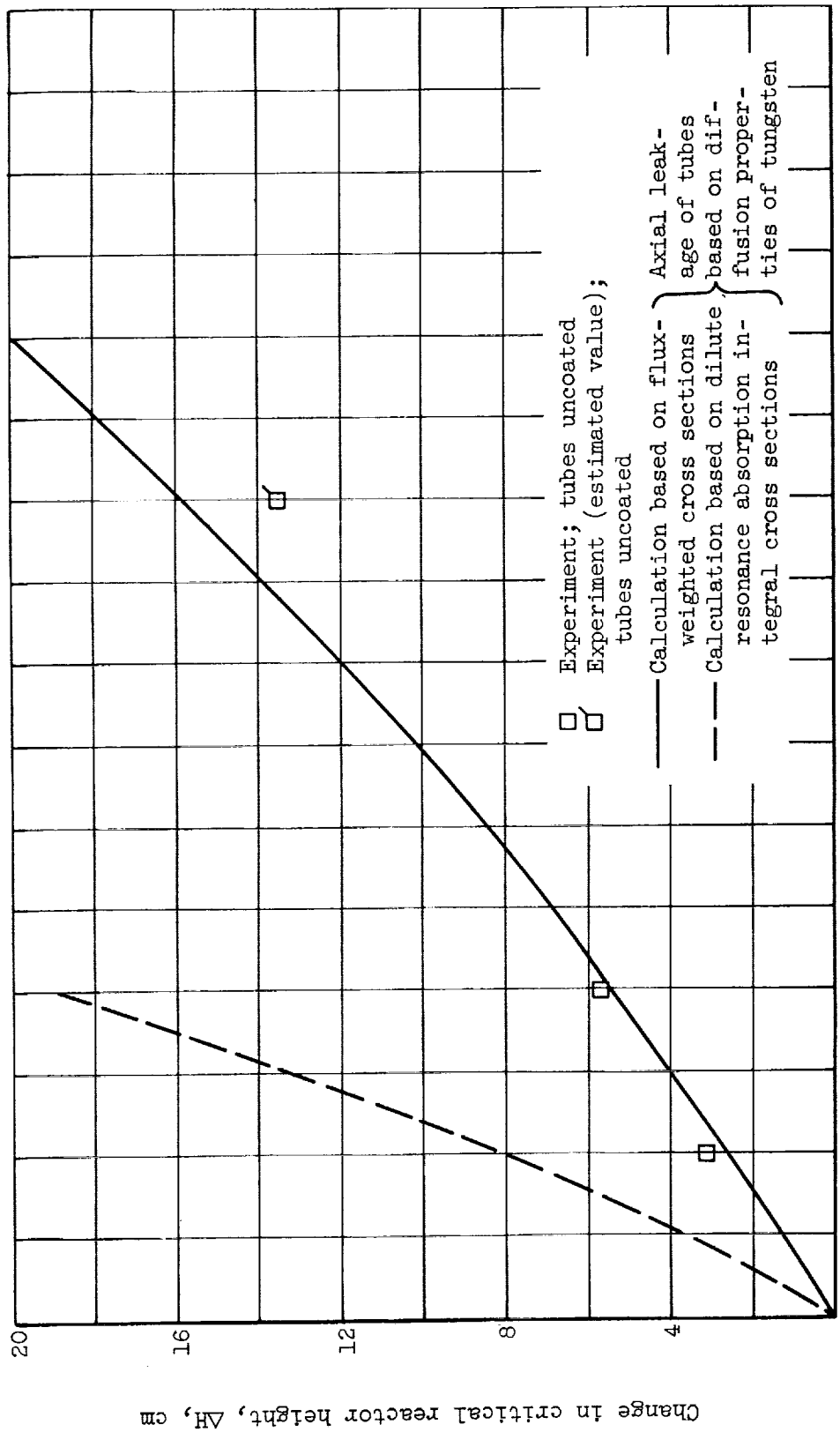


Figure 9. - Change in critical reactor height as function of wall thickness of 4-inch-outside-diameter tungsten-tube inserts. Critical height with no insert, 30.73 centimeters; fuel solution at hydrogen- to uranium-235-atom ratio of 300.

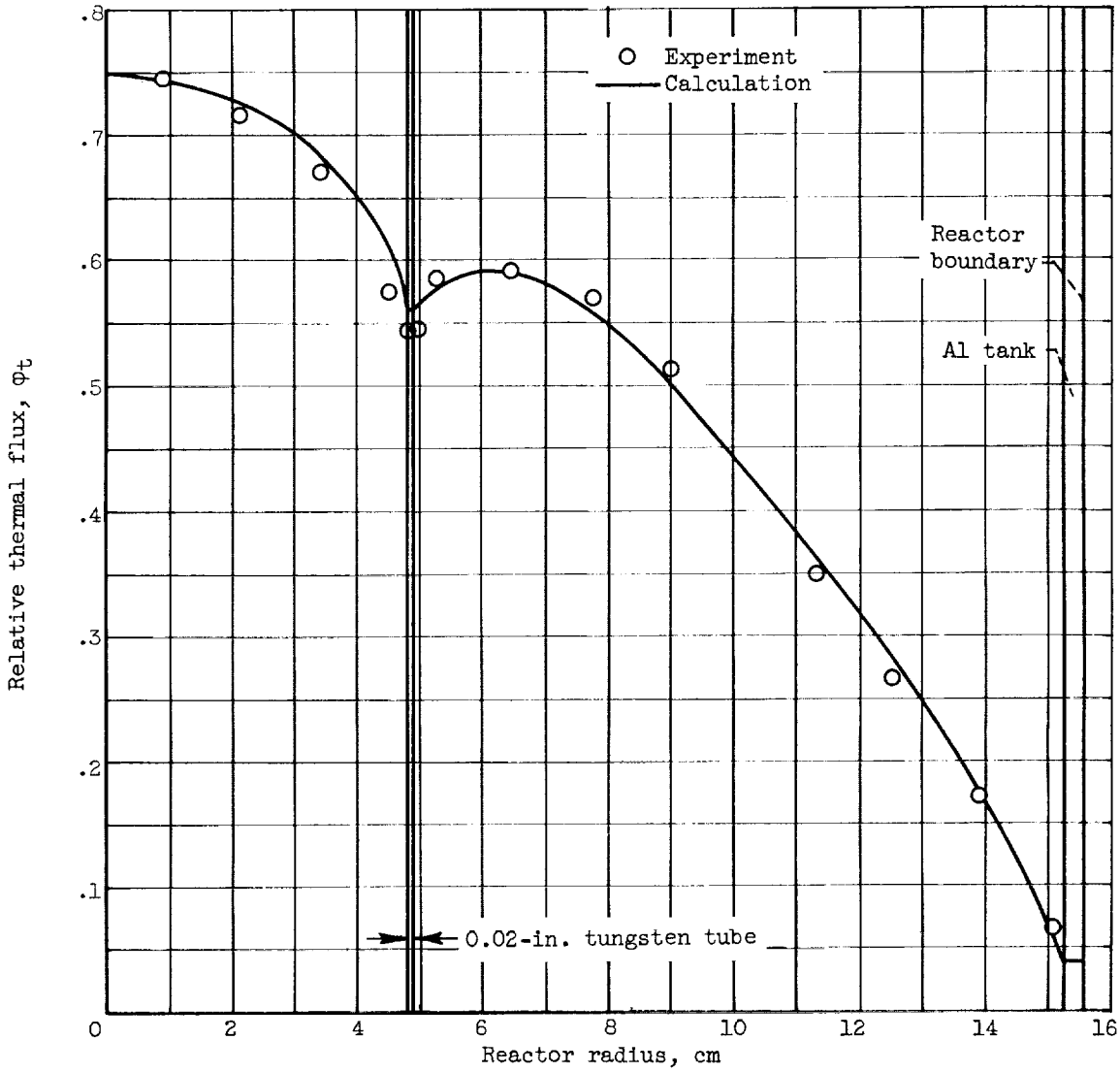


Figure 10. - Relative thermal flux as function of reactor radius with 4-inch-outside-diameter tungsten-tube insert of 0.02-inch wall thickness. Fuel solution at hydrogen- to uranium-235-atom ratio of 300.

NASA TN D-1322
National Aeronautics and Space Administration.
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Daniel Fieno, Eugene Gunn, Clayton Barber, Thomas Fox, Donald Alger, and Robert Mueller.
August 1962. 27p. OTS price, $0.75.
(NASA TECHNICAL NOTE D-1322)

The NASA zero-power reactor, consisting of an unreflected cylinder containing a solution of uranyl fluoride salt in water, was used to study simple heterogeneous effects. This heterogeneity was introduced into the reactor as tubes or rods of aluminum, iron, and tungsten. For a given hydrogen- to uranium-235-atom ratio of the fuel solution, criticality was achieved by varying the height of the fuel solution contained in a 12-inch-inside-diameter tank. Criticality measurements of cylindrical void regions located on the axis of the reactor were also made. These measurements were analyzed by using 19 energy groups with a one-dimensional diffusion-theory code written for the IBM 704 computer.

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